NON-PUBLIC?: N
ACCESSION #: 9312300205
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Joseph M. Farley Nuclear Plant - Unit 2 PAGE: 1 OF 5

DOCKET NUMBER: 05000364

TITLE: Reactor Trip on Steam Generator Low-Low Level following a Turbine Trip
EVENT DATE: 12/02/93 LER #: 93-004-00 REPORT DATE: 12/21/93

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 005

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: R. D. Hill - Nuclear Plant General Manager TELEPHONE: (205) 899-5156

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: JB COMPONENT: PDC MANUFACTURER: W120
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 2233 on December 02, 1993, with the Unit in Mode 1 operating at approximately 5 percent reactor power, the reactor tripped due to a low-low water level in the 2C steam generator (S/G) following a turbine trip and feedwater isolation.

The turbine/generator had been ramped down to between 60 and 70 megawatts in preparation for a turbine overspeed test. At this time reactor power was being maintained at approximately 20 percent power with load being maintained by the main generator in combination with the steam dumps. After the main generator output breakers were opened, S/G level and feedwater flow oscillations occurred. The operators were unable to maintain S/G levels due to the steam generator feed pump speed control maintaining an excessive/erratic pressure differential across the feed regulating bypass valves as a result of a speed controller card failure.

The level in the 2B S/G rose to the point that turbine trip and feedwater isolation signals were generated. With feedwater isolated, the S/G levels decreased. The operators began simultaneously feeding the S/G with auxiliary feedwater at the maximum attainable rate to maintain levels, and began inserting the control rods to reduce reactor power and return the Reactor Coolant System average temperature to the program value. Within approximately 1.5 minutes of the turbine trip/feedwater isolation, the reactor tripped on low-low level in the 2C S/G.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION

Westinghouse - Pressurized Water Reactor.

Energy Industry Identification System codes are identified in the text as XX!

SUMMARY OF EVENT

At 2233 on December 02, 1993, with the Unit in Mode 1 operating at approximately 5 percent reactor power, the reactor tripped due to a low-low water level in the 2C steam generator (S/G) AB! following a turbine trip and feedwater isolation. The turbine trip and feedwater isolation was caused by the level in the 2B S/G reaching the high level setpoint. After the feedwater isolation occurred the operating crew was not able to maintain S/G levels using the auxiliary feedwater system and thus the reactor trip occurred.

DESCRIPTION OF EVENT

On December 02, 1993 during the process of returning the Unit to service after the Cycle 9-10 Refueling Outage, the turbine/generator had been ramped down to between 60 and 70 megawatts (MW) in accordance with unit operating procedures in preparation for performing a turbine overspeed test. Prior to removing the main generator from the grid, the feed-regulating bypass valves (which were controlling the feedwater flow to the S/Gs at the time) were placed in manual so that S/G levels could be maintained high in their operating band due to a S/G flow transmitter being in test. This flow transmitter being in test would cause the reactor to trip at 25 percent S/G level instead of the normal 17 percent on the affected (2B) S/G.

The main generator output breakers were opened at 2206 to remove the main generator from the grid in order to perform the main turbine overspeed trip test. However, when the main generator was removed from the grid, a S/G level and feedwater flow oscillation began that caused the S/G levels to begin fluctuating. At 2232 this fluctuation resulted in a high level in the 2B S/G which caused a main feedwater pump trip and feedwater isolation.

With feedwater isolated, the S/G levels began to decrease. The operators began simultaneously feeding the S/Gs with auxiliary feedwater at the maximum attainable rate to maintain levels, and began inserting the control rods to reduce reactor power and to return the Reactor Coolant System average temperature to the program value. The operators were unable to maintain S/G levels and within one minute of the turbine trip/feedwater isolation, the reactor tripped on low-low level in the 2C S/G.

CAUSE OF EVENT

An investigation into the event revealed that the steam generator feed pump speed control JB! was maintaining an excessive/erratic pressure differential across the feed regulating bypass valves as the result of the failure of a speed controller card. With this differential pressure being maintained above the program value, the feed control system responded erratically to the operator's input during power changes and level control was hampered. Contributing causes were:

1. A mind set among the plant operators to bias high the 2B S/G level to compensate for susceptibility to the 25 percent low level reactor trip.

2. Large load rejection upon opening the main generator output breakers. Operators attempted to reduce load to less than 40 MW per procedure. However, the procedure did not specify to use actual MW in lieu of the demanded MW which was used by the operators.

3. The reluctance of the operators to utilize the control rods to control temperature. Each crew at FNP had received specific training on plant startup with a positive moderator temperature coefficient (MTC). The primary tool utilized to anticipate temperature change with a positive MTC is intermediate range startup rate, However, the MTC for the Unit 2 reload was near zero. With MTC approximately zero, the startup rate was ineffective as a tool to control temperature.

The combination of erratic feedwater flow control, the fact that the feed regulating bypass valves were being operated in manual, and the fact that the 2B S/G level had been fed to the high end of its band (a planned compensatory measure for the S/G flow transmitter that was in test) all contributed to the feedwater isolation and subsequent reactor trip.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable because of the actuation of the reactor protection system. All systems operated as designed in response to the reactor trip.

CORRECTIVE ACTION

A root-cause analysis investigation was performed. This analysis included interviewing all of the operating crew that was involved in the trip as well as establishing in detail the exact sequence of events and each crew member's actions during the event. The speed control problem was determined to be the major contributing cause and this was corrected by replacement of a controller card in the system prior to synchronizing the Unit to the grid.

The S/G flow transmitter that had been in test when the trip occurred was returned to service.

The main generator was ramped to a lower power level prior to removal from the grid, before attempting the main turbine overspeed test again.

The event and its causes were reviewed with each operations crew that would potentially be involved in performing the subsequent testing.

Each operations crew will receive specific training on this event.

The following procedure enhancements have been made:

a. A caution regarding temperature control with zero MTC has been added.

b. Specific instructions to reduce actual MW has been added.

C. Specific instructions have been added to verify proper feed pump speed control prior to power escalation.

ADDITIONAL INFORMATION

A similar event was reported for Unit 2 on 06-11-92 per LER-92-005-00, "Reactor Trip on Low Steam Generator Level Coincident with Feedwater Flow less than Steam Flow Signal"

The Reactor was taken critical following the trip on December 03, 1993 at 0602 and the main generator was synchronized to the grid at 2359.

No equipment (other than the above mentioned feed pump speed controller card) failed during this event and the event would not have been more severe if it had occurred under different operating conditions.

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Dave Morey Southern Nuclear Operating Company
Vice President the southern electric system
Farley Project
December 21, 1993

10 CFR 50.73

Docket No. 50-364

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Joseph M. Farley Nuclear Plant - Unit 2
Licensee Event Report No. 93-004-00

Gentlemen:

Joseph M. Farley Nuclear Plant, Unit 2, Licensee Event Report No. LER 93-004-00 is being submitted in accordance with 10 CFR 50.73 . If you have any questions, please advise.

Respectfully submitted,

Dave Morey

DPH/sar:LER-004.DPH

Enclosure

cc: Mr. S. D. Ebneter
Mr. B. L. Siegel
Mr. T. M. Ross

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